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5-YEAR OPERATION EXPERIENCE ON REACTOR BR-5

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The employment of fast-neutron breeder-reactors in atomic power engineering opens up broad avenues for use of entire natural uranium and thorium as nuclear fuel.The designing and construction of industrial atomic power stations with fast-neutron reactors required the answer to many specific engineering and technical features of running reactors of such type. The problem of building a small power research fast-neutron reactor with parameters approaching those of future industrial installations arose in the Soviet Union in 1956.

During the running of this particular reactor it was necessary to prove the practical feasibility of the ideas to build fast-neutron reactors using liquid-metal coolants, to gather experience of work with liquid metals sodium and sodium-potassium alloy in a nuclear installation, to obtain experience of work with fuel elements under the deep burnup of the nuclear fuel, to examine the kinetic characteristics of a fast reactor with core high power density, to acquire experience of operation of individual elements of the technological and reactor equipment, control instruments and so forth. The pilot fast-neutron sodium-colled reactor BR-5 was developed in the Soviet Union in 1957-58 for these purposes.

The description of the BR-5 reactor has been published (1) and is not given in this report.

The five-year period of reactor operation can be divided into several stages depending on the tasks tackled on the installation.

Stage 1 – the start-up and adjustment and study of the reactor's physical characteristics. (December 1958 – June 1959).

This stage covered the start-up and adjustment of the liquid-metal coolants of the installation, the physical start-up of the pile with the coolant in January 1959 and studies of the reactor's physical characteristics at no more than 20% nominal capacity.

Stage 2 – the operation of the reactor to reach the designed fuel burnup 2% (July 1959 – October 1960). The designed power of the pile of 5,000kws was reached on July 21 1959, and in the indicated period the pile operated at various power levels including the nominal coolant outlet temperature 400–450°C. This stage covered the working up of all elements of the technological and reactor equipment and studies of the steady and transient rates of the coolant natural circulation in circuits and loops during power operation of the reactor and investigation of the pile's power dynamic characteristics.

Stage 3 – reactor operation to reach maximum possible fuel burnup. (November 1960 – September 1961). The reliable operation of the reactor fuel elements till the designed fuel burnup was reached made it possible to decide to carry on running the pile without replacing the fuel elements in the core until there were sure signs that the fuel elements were out of commission. Simultaneously, in December 1960 the outlet temperature was raised to 500°C, since it was believed expedient to bring the temperature level of the technological research on the BR-5 reactor to the temperature conditions on industrial plants of such type. By January 1, 1961 the reactor had generated 21.10^6kw-hr. The maximum of 2.8% burnup was reached in the fuel elements. In the course of 1961 the pile worked the bulk of the time at the nominal power 5,000kws with sodium outlet temperature 500°C. In 1960-61 the pile was in operation for more than 80% of time. At this stage the pile was used to carry on the nuclear physics research program, to irradiate samples of different fuel elements and design materials both in the reactor core and in the reflector. By September 1, 1961, the pile had generated 39. 10^6kw-hr and a maximum of 5% of fuel was burnt up. The integral dose of fuel element exposure in the center of the core was $2.2x10^{22}n/cm^2$. The pile was stopped in September 1961 to examine the state of the fuel elements because of considerable contamination of the coolant and the primary circuit equipment with fission products.

Stage 4 – reactor operation while there have been untight fuel elements in the core. (From March 1962 on).

After the state of the fuel elements had been examined and the primary circuit decontaminated of fission product in March 1962, the BR-5 reactor was put into operation again. 80% subassemblies of plutonium dioxide were loaded repeatedly into the pile for further work. The aim of this stage of reactor operation was to study the release of solid and gaseous fission products from the untight fuel elements into the coolant and the gas blanket of the primary circuit depending on the temperature and power rates of reactor operation. The task was tackled of studying and identifying by components the deposition of radioactive fission products on the walls of the technological equipment and pipe lines of the primary circuit of the installation of fission product.

In the course of 1962 and 63 the pile operated at various power levels depending on the needs of the particular experiment with the sodium outlet temperature 300–430°C. The maximum burnup reached in the plutonium dioxide fuel elements by March 1, 1964, was 5.8%.

START-UP AND ADJUSTMENT OF LIQUID-METAL REACTOR CIRCUITS AND REACTOR OPERATION EXPERIENCE BEFORE DESIGNED BURNUP IS REACHED

The work to start up and adjust the main reactor circuit included the following measures for each circuit:

1. Drying and degassing of the circuit.
2. Washing of the circuit and the running down of the circulation pumps.
3. Running down of the circuit with the working coolant and checking of the hydraulic characteristics at the same time.

Drying and degassing of liquid-metal reactor circuits was done by vacuuming till satisfactory readings of extreme vacuum and inleakage were obtained. The primary circuit was both vacuumed and warmed by inbuilt electric heating and gas heating of the central reactor tube. The extreme

vacuum reached in the circuits was 1.10^{-1}mm of the mercury column and the inleakage was $2-4.10^{-2}$mm of the mercury column per hour. After they were dried,the circuits were filled with purified argon and an argon overpressure was then maintained in them. The washing of the circuits was done with pre-distilled eutectic alloy Na-K by starting the alloy circulation with inbuilt pumps, heating up to a temperature 300°C with the circuit electric heating and periodic draining of the alloy with subsequent cooling down to a temperature of 50–100°C. After washing with the alloy the primary circuit was filled with sodium coolant pre-distilled in a special installation. The impurity content in the distilled sodium was: oxygen $<2.10^{-3}$%, carbon $<2.10^{-3}$%, hydrogen $<9.10^{-3}$%.

The content of oxides in sodium after starting the circulation in the primary circuit did not exceed 3.10^{-3}% in oxygen. Such presence of oxides was then maintained with a cold trap.

During the running down of the primary circuit the effect of gas transfer with the coolant was detected. Gas was pumped from the gas blanket of the central reactor tube. This was detected by the change in the sodium levels. Simultaneously, part of the pumped gas strayed in the pipelines and produced increased resistance of the circuit. This entailed lower flow rates of sodium in the circuit. The pumping of gas was taking place because the pump during its operation formed a gasometallic emulsion and the working wheel of the pump caught it. After several changes were made in the design of the circulation pump in order to ensure organized flow of the coolant to the working wheel of the pump and to create a quite-mirror sodium in the pump tank there was no significant transfer of gas in the circuit even when the pumps worked for a long time.

However, when the physical characteristics of the pile were studied after the reactor was started up physically with the coolant, an effect was found that reactivity changed when the pressure in the central reactor tube gas blanket changed as well as in the process of sodium circulation through the primary circuit. A series of special experiments showed that the change in reactivity was linked with accumulation of gas in the reactor core in the form of the smallest bubbles depositing themselves on the fuel element surfaces. A number of measures that were described in detail in paper (2) made it possible to reduce the effect to a minimum. The start-up adjustment showed that the operation of small emergency pumps, installed on one loop of the primary and secondary circuits for ensuring emergency cooling down of the pile in case the external power supply sources are cut out completely, in parallel to the main circulation pumps causes instability in the maintenance of the levels of the coolant and pressure in the gas blankets of the pump tanks. For this reason the emergency pumps were taken out of the primary and secondary circuits. To ensure emergency cooling down of the pile within the first minutes after the external power supply source was cut out, when the residual heat release is still great because of delayed neutrons the circulation pump of one of the primary circuit loops was connected to the "steady supply" main. The heat removal from the primary circuit was carried out by natural circulation of the alloy through the corresponding loop of the secondary circuit and by natural air draught through the air heat exchanger.

Special studies of the stationary and non-stationary rates in the natural circulation showed that the air loop ensures fool-proof heat removal with the help of natural circulation of up to 250 kw power which ensures quite well the removal of residual heat release in the pile in several minutes after it is stopped.

At the opening stage of operation when the pile worked still on a low power level, two cases were recorded when wire electric heaters burnt first the pulse tube and then the drainage pipeline of the primary circuit. In order to prevent such developments an electric heating supply diagram was carried out on the installation for the circuits through the separating transformer, which made it possible to rule out burning through of the pipelines when the electric heaters shorted.

In October 1960 the steam generator went out of commission because of ruptures in several tubes dividing the mercury and distillate blankets and in one tube between mercury and Na-K alloy. By that moment the steam generator had worked a total of 4200hrs.

The cause of mercury-water tubes disintegrating was chlorine corrosion of the stainless steel by the water in the evaporation part of the steam generator. The mercury-Na-K tube burst in the spot where it was welded to the tube board of the steam generator, evidently because of a factory defect. Before a steam generator of a new design was developed and manufactured an air heat exchanger was assembled on the second loop of the secondary circuit, similar to the Na-K-air heat exchanger installed on the first loop.

REACTOR OPERATION AFTER DESIGNED FUEL BURNUP WAS REACHED

During the first two stages of the BR-5 reactor operation (from January 1959 to September 1, 1961) the control of the coolant activation was done by means of measuring the nature of the γ-activity decay in the boxes of the primary circuit after the pile was stopped as well as by means of chemical and γ-spectrometric tests of coolant and gas samples periodically taken from the primary circuit and its gas blankets.

The first signs indicating certain disturbance in the fuel element density appeared when the designed (2%) plutonium burnup was reached. At the first stage the coolant activity diminished 100,000 times to the level determined by the activity of sodium-22 with a half life of T 1/2≅15hr. Beginning from the moment when the plutonium burnup reached 2.4% residual primary circuit activity was registered, which increased as the fuel burnup progressed (See Fig.1). Also during this stage no fission products or γ-activity were detected in the selected sodium and gas samples, which indicated the tightness of the fuel element canning. In October 1960 Xe-133 was spotted in the central reactor tube gas blanket. This showed that the tightness of individual fuel elements had been disturbed. Absence of α-activity and solid fission products in the primary circuit proved that the degree of damage to the fuel elements was small. In the coolant samples taken from the circuit when a 3.2% burnup was reached Cs-137 was detected along with the Na-22 isotope. The Cs-137 activity contribution changed as the burnup progressed in the following way (See Table 1).

On September 1, 1961, when a 5% fuel burnup was reached, the pile was stopped because a primary circuit circulation pump had gone out of commission. After a two-week exposure, the γ-background in the primary circuit amounted to several thousand micro-R per sec. The drainage of the coolant from the circuit did not diminish the activity of the equipment and the pipelines of the primary circuit. This indicated that the bulk of active impurities had deposited themselves on the circuit's inner walls. The coolant residual activity was determined mainly by the Cs-137 isotope. In addition, α-activity (of plutonium) was detected in the coolant with an intensity of 50 decays per sec. lgr of sodium.

Table 1

Fuel Burnup %	Residual Activity induced by Na-22 %	Residual Activity induced by Cs-137 %	α-activity (decay per sec. per sodium 1 gr.) %
2.00	100	–	–
3.20	80	20	–
3.90	60	40	–
4.55	30	70	–
5.00	5	95	50

All this pointed to a noticeable disturbance in the canning of some fuel elements, which caused washing out of the fission material.

Since it was necessary to examine the tightness of the fuel element canning in October 1961 the reactor core was completely unloaded. All 120 subassemblies of the core and the shield were removed into the storage volume with the help of the unloading container.

Part of the subassemblies were processed with steam in a special system for washing off sodium remains from subassemblies before the fuel element canning were examined for tightness, but since the fuel material was washed away from some untight fuel elements the steam treatment was stopped.

Both the steam processed and non-processed subassemblies were checked for fuel element canning tightness. The measuring of the activity of the gas pumped off from the subassemblies showed that 18 out of 81 subassemblies produced an activity thousands of times greater.

These subassemblies were discarded. Two of the discarded subassemblies with a 4.9% burnup (one was processed with steam) and one subassembly with a 5% burnup out of the undiscarded were examined in the hot laboratory. Longitudinal cracks 8 – 10 cm long and up to 1.0 mm wide were detected on the surface of some elements of these subassemblies. The cracks, as a rule, continued in the plutonium dioxide. Mostly radial cracks were observed on the surface of cross sections of the disintegrated elements. No central cavities were found in any of the disintegrated elements. In the third subassembly examined all the elements turned out to be tight, the element canning was in good condition: the element surface was light-colored and bright and there were no visible defects.

The primary reactor circuit was processed with steam the circuit electric heating cut in after the core was completely unloaded and sodium drained into a special storage place. No claps were observed in the steam treatment of the circuit. The degree of γ-radiation dose in the boxes where the primary circuit equipment is housed diminished by about one-half after the treatment. The technological tract of the primary circuit was then filled in the following order: with a 0.5%$KMnO_4$ solution, the mixture of a 5% nitric acid solution and 1% oxalic acid solution and with distillate. After several cycles of processing the primary circuit with these solutions the γ-background in the boxes decreased to 10 micro-R per sec. A radiochemical test of the drained solution from the primary circuit showed that the γ-activity of the solutions was caused mainly by Cs-137, Zr-95 and Nb-95.

The final stage of decontaminating the primary circuit was distillate washing to remove the remains of the acid. The circuit was dried after washing by vacuuming accompanied by heating the equipment and pipelines up to a temperature of ~150°C.

To examine the operation capability of the oxide fuel elements at a more than 5% burnup and to examine the contamination with fission products of the coolant and the primary circuit tract it was decided to load for the new period of operation more than a half of the subassemblies that were in operation.

The core was loaded with subassemblies used in the pile for a second time by batches of 15–20 subassemblies. The subassemblies had undergone no advance treatment to remove sodium oxides. Therefore after each batch of subassemblies was loaded circulation of sodium through the core was cut in at full flow rate to dissolve oxides from the subassemblies and to transfer them into cold trap. After sodium oxides were dissolved from the subassemblies completely, which was gauged by the increase of sodium flow rate through the core up to the nominal one, another batch of 15–20 subassemblies was loaded. The core was loaded with a total of 63 plutonium fuel element subassemblies that had worked for three years and with 50 new subassemblies with the fuel in the form of uranium dioxide.

EXAMINATION OF THE RADIOACTIVE IMPURITIES IN PRIMARY CIRCUIT COOLANT AND INERT GAS IN 1962–63

To examine the contamination of the coolant and circuit with fission products a special bypass section of the primary circuit with a diameter of 40 mm was mounted in March 1962 (See Fig. 2). The section is cut off from the circuit with two valves, which make it possible to set sodium flow rate up to 5.5 m^3/hr, and has a system of emptying independent of the circuit. Part of the section in a lead shield leads into the serviced premises. A collimated flux of γ-rays falls from the pipeline into the NaI(TL) crystal 40 by 40 mm of the transducer of the 100-canal scintillation γ-spectrometer. Sodium was circulated in the section to examine the radioactive impurities in the circuit while the pile worked at a given operating codition. The conditions of the experiment set the circulation time, the sodium flow rate in the section and the operation of the section electric heating. After the circulation was stopped, sodium was frozen in the section and in 8 to 9 days (after the decay of the Na^{24} isotope) measurements were taken of the γ-ray spectrums of the section filled with coolant.

When the pile operated at 1,000kw power with the sodium outlet temperature 350°C the residual γ-activity of the coolant was determined by the fission products Cs^{137}, Ba^{140}, Zr^{95}, Nb^{95} and $Na^{22}La^{140}$ isotope. 90% of the coolant's residual γ-activity was caused by isotope Cs^{137} and 5% by isotope Na^{22}. The Cs activity in the coolant held on at a steady $\sim 5.10^5$ decays/m/lgr of sodium. The oxide filtration system worked more than 65% of the entire time during this period. The citting out of the filtration system for 24 days did not produce any noticeable change in the Cs activity in the coolant. Repeated measurements showed that the activity of the section of the pipeline after it had been emptied decreased only by 10–20% despite the fact that sodium is drained practically completely from the section, which is indicated by the decrease in the Na^{22} isotope activity by hundreds of times.

A similar phenomenon was observed when sodium was drained from the loops of the primary circuit, if the draining was done 8 or more days after the pile was stopped. The degree of radiation in the box of the loop drained practically did not diminish. This indicated again that a

sizable part of radioactive fission products deposited itself on the inner surfaces of the pipelines and equipment of the primary circuit. Since the process of radioactive impurity deposition on the inner surfaces of the primary circuit pipelines and equipment and the process of washing the impurities from the walls are of great practical interest, a number of tests were made to study the influence of different technological operating conditions of the primary circuit and the bypass section on the concentration of Cs^{137} on the inner walls of the pipelines.

It was established that the change in the temperature of sodium in the primary circuit and the change in the sodium flow rate through the section did not result in a noticeable change in the concentration of Cs on the inner walls of the tube.

It was noticed that when electric heating of the spectrometer section was cut in with the coolant circulating, the concentration of Cs^{137} on the inner walls of the section diminished by several times, and when the heating was cut out it gradually picked up. To study this phenomenon and to examine the effect of oxide cold trap operation on the concentration of Cs in the coolant, a number of experiments were staged in January 1963 on the dead pile (after the decomposition of Na^{24}). It was established that with the filtration system cut off when the electric heating of the pipelines and the equipment of the primary circuit was cut in the Cs^{137} concentration in the coolant increased several times. After the filtration system was cut in, Cs concentration quickly diminished to the initial level. In the balanced state the Cs content in the coolant depends on the cold trap operating conditions: the higher the temperature of sodium inside the trap, the higher the concentration of Cs in the primary coolant. When the temperature inside the trap was increased from 110°C to 185°C, the Cs activity in the coolant jumped by three times. When the pile operated at 2,000 kw power with the filtration system cut off the Cs activity in the coolant doubled within ten days and the activity of Ba^{140}, La^{140}, Zr^{95} and Nb^{95} increased sizable. I^{131} was also detected in the coolant with an activity three times as great as the Cs activity. The cutting in the filtration system diminished the Cs^{137} concentration to the former level and the I^{131} concentration dropped noticeably. So, the studies made at powers of 1,000 and 2,000 kw showed that the increase in the coolant temperature did not result in a significant increase in the fission products outflow into the coolant while the increase of the power and consequently in the fuel temperature increased the outflow of fission products into the coolant.

In addition to studies of the primary coolant, spectrometric studies were made of the γ-activity of various parts of the equipment extracted from the primary circuit (transducers of the pump level gauges, the pump working surfaces and samples of the pipeline cut away from the primary circuit).

The activity of the working surfaces of the pump (unwashed) and the level gauge (washed from sodium) removed from the circuit during the 1,000 kw operation was caused by Cs^{137} isotope both in the region submerged in sodium and in the region above the sodium level in the gas blanket. Besides the 0,66Mev line (Cs^{137}) in the pump and level gauge in γ-ray spectrum there were noticed the lines 0.8 Mev and 1.6 Mev ($Ba^{140} - La^{140}$ and $Zr^{95} - Nb^{95}$). The Cs activity is at the maximum on the borderline between sodium and gas (See Fig. 3). The activity of the working surfaces of the level gauges removed from the circuit after operation at 2,000 kw before they are washed from sodium was caused by Cs^{137}, $Ba^{140} - La^{140}$ and $Zr^{95} - Nb^{95}$ (See Fig. 4).

After the level gauges were washed with steam and water the Cs activity diminished by ten times, the activity of $Ba^{140} - La^{140}$ by one-third, and the activity of $Zr^{95} - Nb^{95}$ decreased insignificantly (See Fig. 5). After washing the Cs activity is determinate only in the region of the level gauge located above the sodium level. In the region submerged in sodium the level gauge activity is determined by isotopes $Ba^{140} - La^{140}$ and $Zr^{95} - Nb^{95}$. The spectrometric studies of pipe samples cut away from the primary circuit in September 1963 and January 1964 showed that both before and after washing the activity of the pipe inner surfaces is determined by isotopes $Ba^{140} - La^{140}$, $Zr^{95} - Nb^{95}$ and Cs^{137}.

To examine the composition of radioactive impurities of the gas from the gas blanket of the reactor a sampling system was mounted in October 1962 that made it possible to sample gas when the pile operated at power. The system is cut into the reactor's central tube gas line and into the gas line of a primary circuit pump. Before sampling gas is blown from the stagnant section of the gas line into a special ventilation, after which gas is sampled into a special tank with a volume of ~1,100 cm^3 for spectrometric studies. To identify the admixed radioactive isotopes of the inert gas γ-ray and half lives were measured with a scintillation γ-spectrometer.

The measurement showed that in addition to isotope Ar^{41} the inert gas had a number of radioactive isotopes listed in table 2.

Table 2

THE RESULTS OF MEASUREMENTS OF γ-RAY ENERGY AND HALF LIFE OF THE GASEOUS FISSION PRODUCTS FROM THE REACTOR BR-5 GAS BLANKET

Measurement		Identified as isotope	Isotope characteristics	
γ-ray energy (Mev)	Half life		γ-ray energy (Mev)	Half life
0.08	5.6 days	Xe^{133}	0.081	5.3 days
0.040 0.18	8.3 days	Xe^{129m}	0.040 0.196	8.0 days
0.24 0.605	9.3 hours 10.0 hours	Xe^{135} Cs^{135}	 0.25 0.605	9.13 hours
0.24	56 hours	Xe^{133}	0.233	55.2 hours
0.51	Very great	$Kr^{85}Rb^{85}$	0.517	10.6 years
0.87 1.55 1.9 2.45	6 hours 3.15 hours 3.8 hours 3 hours	$Kr^{85}Rb^{88}$	0.91 1.85 2.4 2.7	2.9 hours

The isotopes that cause the gas activity are Xe^{133} and Xe^{135}. As an example Table 3 features the activity of various isotopes of the gas sampled from the reactor gas blanket at 1,000 kw power on February 19, 1964.

Table 3

No.	Isotope	Activity (μC per cm^3)
1	Xe^{133}	4.3
	Xe^{135}	0.95
	Xe^{133m}	0.05
	Kr^{85}	0.017

It is noteworthy that in none of the samples of the gas were there detected short-lived fission products with a half life less than 2.5 hours although the sampling and measuring methods made it possible to detect isotopes with a 5-minute half life.

KINETIC STUDIES

The examination of the reactor BR-5 kinetics is important both for the reactor operation and for the understanding of phenomena taking place in a reactor of such type. The asymptotic power effect, determined with a constant inlet sodium temperature was measured by the change in the position of adjustment organs, that ensure the reactor criticality at various powers. The asymptotic power coefficient at the beginning of the reactor operation was 4.5×10^{-4} l/mw and diminished in the process of operation to $(3-1).10^{-4}$l/mw. At the same time the value of the power effect was measured, that takes place 5–50 seconds following a power change. While at the beginning of operation the effect was equal to zero, it became positive and reached $(2-4).\ 10^{-4}$l/mw at a 2.5% burnup. The explanation of the phenomenon was connected with the "swelling" of the fuel material, its adhesion to the canning and the decrease of the fuel element elongation because the heated canning much less than the plutonium oxide did. After part of the fuel in the core was replaced with fresh uranium oxide the measurements of power effects were repeated by two methods.

In the first case a step-function signal was put into the reactor automatic control system. The power changed to a new stationary value within 1 – 1.5 seconds. The AR rod position change required for this was fixed with a recorder. The effect of change on reactivity was determined as the difference in the rod behavior at great and small power. The results of measurements at various coolant flow rates normalized to the constant change in sodium heating (+ 70°C) are represented on Fig. 6. Within the first seconds no feedback reactivity change was detected, which is connected with a shortcoming of the method. During this time all fuel elements of the core are heated with a time constant of 1 – 2 sec. The fuel heating contribution to reactivity must change in passing to small flow rates while the rest of the processes practically retain their values. Within a period of up to 20 seconds the central tube housing the core is heated up. This causes a drop in the core and its shifting with regards to the reflector adjustment elements. How positive reactivity appears as a result of this phenomenon is seen clearly at low sodium flow rates. After 20–25 seconds at flow rates of 245m^3/hr and 145m^3/hr the feedback reactivity diminished because the first portions of heated sodium return to the reactor inlet. At lesser flow rates this phenomenon sets in later. In the period of 5–20 seconds at 245m^3/hr the zero value of the power effect was again obtained just as it was during the opening

reactor operation. Apparently, this was again caused by the elongation of the oxide column in the new fuel elements. The observed dependence of the asymptotic value of the first two groups of effects on the sodium flow rates makes it possible to determine the influence of fuel heating on reactivity. Fig. 7 represents this dependence (reactivities are renormalized for the constant power change of 5 mw) as the function of reciprocal flow rate. The extrapolation $1/G \to 0$ gives the fuel effect value $\Delta\rho_{fuel} = -(0.075 \pm 0.015)\ \frac{\%}{5mw}$.

The possibility of normalizing the experimental data on the equal change in sodium heating or the equal change of power needs experimental confirmation of the linearity of the processes. Such an experiment at a flow rate of 28 m^3/hr at various powers with heating of 30 – 110°C showed that in the range of 0 – 70°C the linearity takes place. However, at great heating non-linearity occurs and the asymptotic value of the reactivity of the first two groups of effects becomes negative. The mechanism of non-linearity is apparently connected with the disappearance of gaps. The oscillation method was also employed to study the power effects. In this case a harmonic signal was introduced into the reactor automatic control system. The AR rod performed nearly harmonic oscillations and produced power oscillations. A special harmonic analyzer isolated the amplitude and phase (with regards to the master oscillator) of the first harmonic in these oscillations. The results of reactor amplitude and phase characteristic measurements at a sodium flow rate of 245 m^3/hr and at powers 500 and 2,000 kw are represented on Fig. 8. The transient characteristics built on the results of these preliminary oscillator measurements has a negative value during the first seconds, confirming the presence of a negative fuel effect, which was not observed directly in the first method. In the period of 10–50 seconds the transient characteristics obtained by the two methods coincide. However, to obtain reliable quantity measurements the oscillation method with the analyzer requires improved analyzer operation. Despite the fact that the pile had a positive component in power effect at lesser coolant flow rates and with great fuel burnup at nominal rate, the operation was nominal at automatic reactor control.

MAIN RESULTS OF FIVE-YEAR REACTOR OPERATION

The five-year operation of the BR-5 reactor has confirmed practical feasibility of industrial installations of such type.

The plutonium dioxide fuel elements have showed good operation capability at a coolant temperature of 500°C and a stainless-steel canning temperature of 600°C. The 5.3% fuel burnup reached, which is three times greater than the designed value, gives ground to believe in good radiation resistance of the fuel elements of such type.

All the designed units of the installation and the equipment and technological control instruments have been operating safely the start-up and improvement. The emergency cooling down system of the reactor with the help of natural coolant circulation in the main and intermediate circuits of the installation is very convenient and reliable. The results of experimental investigations of natural circulation rates have given good coincidence with the calculated data. Alkaline metal oxide cold traps have ensured maintenance of oxygen concentration in sodium

and Na-Ka alloy within limits of 0.001 - 0.005%. During the five years of operation cold traps were replaced four times in the primary circuit and once in the intermediate circuits. The fuel reloading system with moveable plugs and transport technological appliances used on the installation has ensured execution of the necessary reloading operations for individual loops of fuel elements and the entire reactor core. The operation duration of the main circulation pumps of the sodium and Na-Ka alloy circuits without replacement has averaged 8,000 hrs. The need to replace the pumps appeared mainly because of defects in the roll bearings.

The exploration at various power levels of the kinetic characteristics of the reactor has shown the nuclear stability was enough for safe operation in conditions of positive power effects.

The presence of solid and gaseous fuel fission products in the coolant and gas of the primary circuit has produced no special complications of the reactor operation. Investigations have shown that sodium-oxide cold traps are also traps for catching solid fuel fission fragments.

In 1964 the BR-5 reactor will continue to be used for studies of the operation of the reactor with untight fuel elements out of mixed oxide plutonium-uranium composition in order to obtain more exhaustive information about the release of solid and gaseous radioactive fission products from the fuel elements into the technological circuit. In late 1964 and early 1965 a transition is planned to new loading of the reactor with fuel out of uranium monocarbide. At present new loading fuel elements have been developed and manufactured. 5 packages of uranium monocarbide fuel have already been loaded into the pile for studies.

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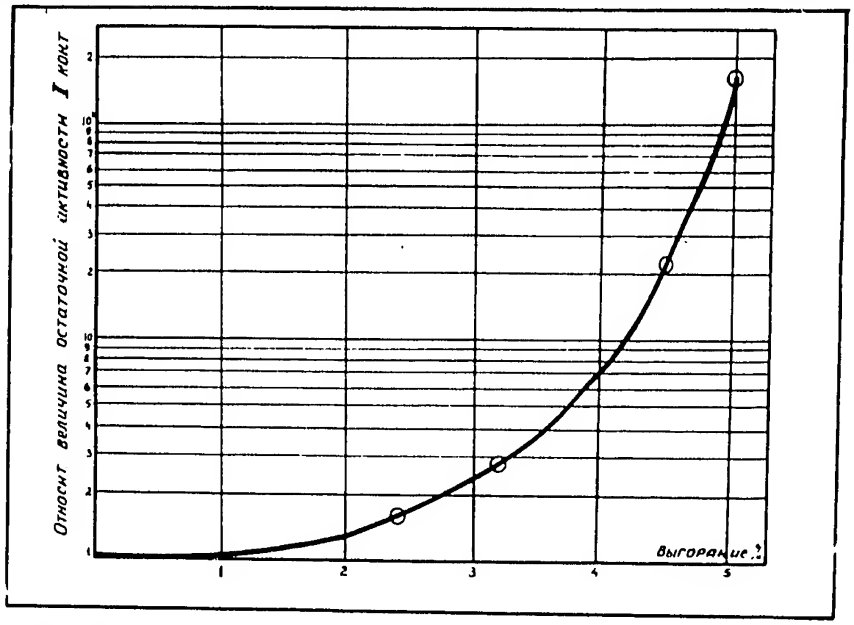


Fig.1 The growth of the residual activity of the primary circuit depending on fuel material burnup.
1.The relative value of the residual activity of the primary circuit.
2.Burnup %.

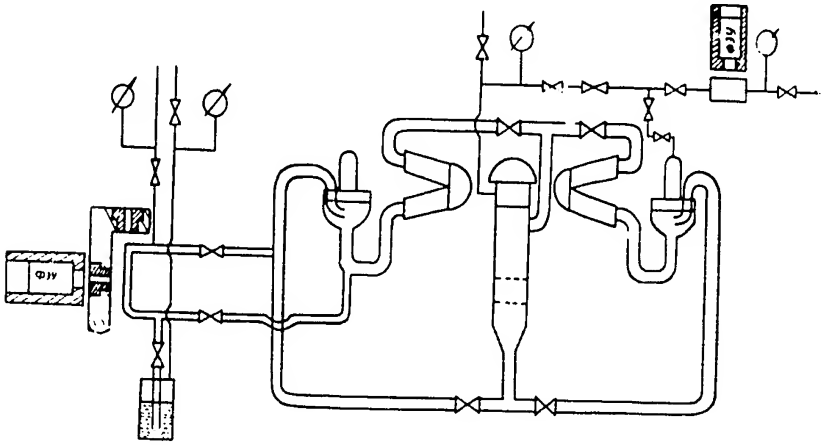


Fig.2 Diagram of the reactor BR-5 primary circuit with a bypass section and gas sampling system.

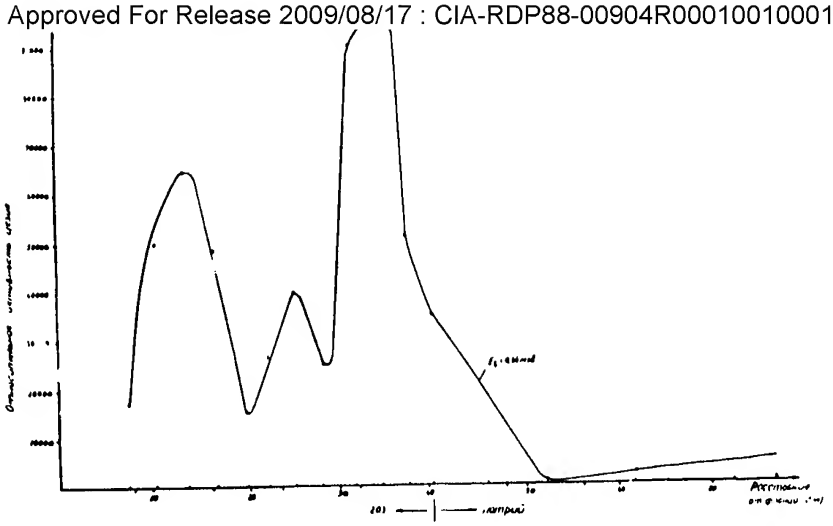


Fig.3 Distribution of Cs-137 activity along the level gauge washed from sodium after it was removed from the primary circuit in April 1963.
1.The relative activity of the cesium.
2.The distance from the flange (cm).
3.Sodium.

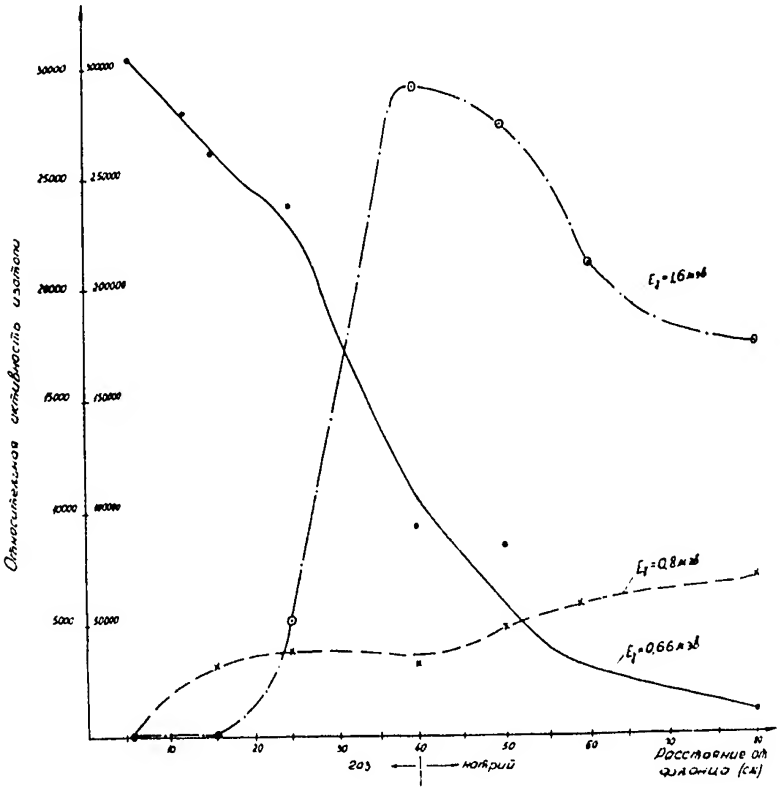


Fig.4Distribution of the activity of Cs-137,Ba-140,La-140 and Zr-95 - Nb-95 along the leval gauge unwashed from sodium after it was removed from the primary circuit in January 1964.
1.The relative activity of the isotope.

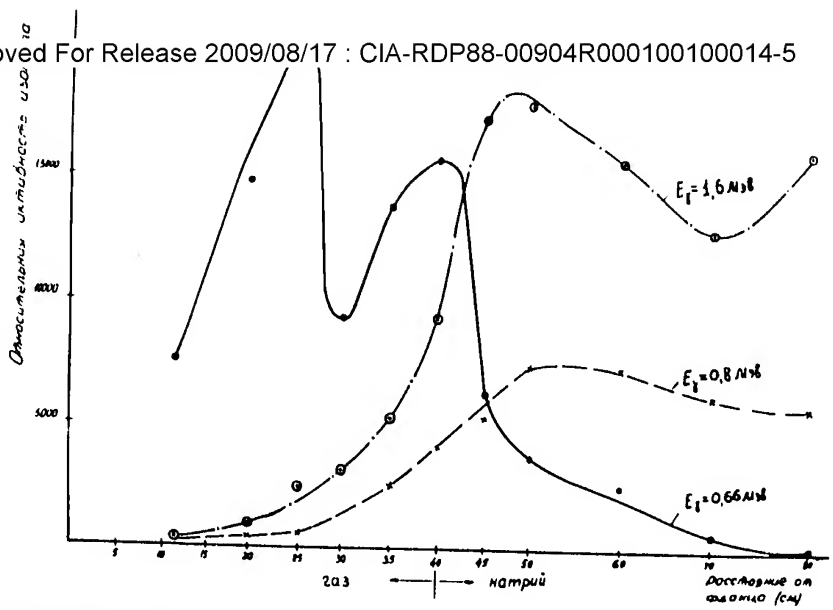


Fig.5 Distribution of the activity of Cs-137, Ba-140, La-140 and Zr-95 - Nb-95 along the level gauge removed from the primary circuit in January 1964 after it was washed from sodium.
1. The relative activity of the isotope.
2. The distance from the flange (cm).
3. Sodium.

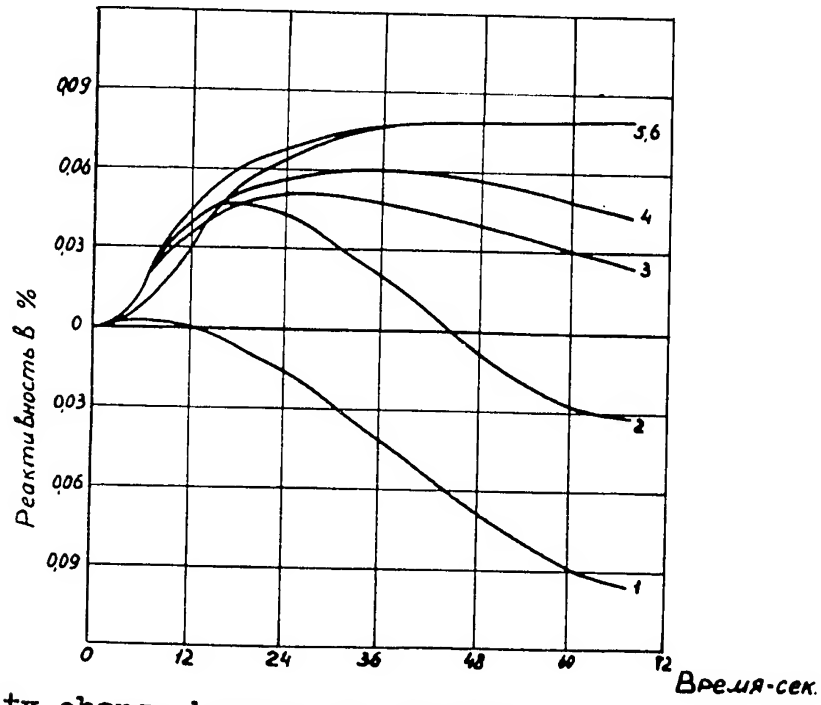
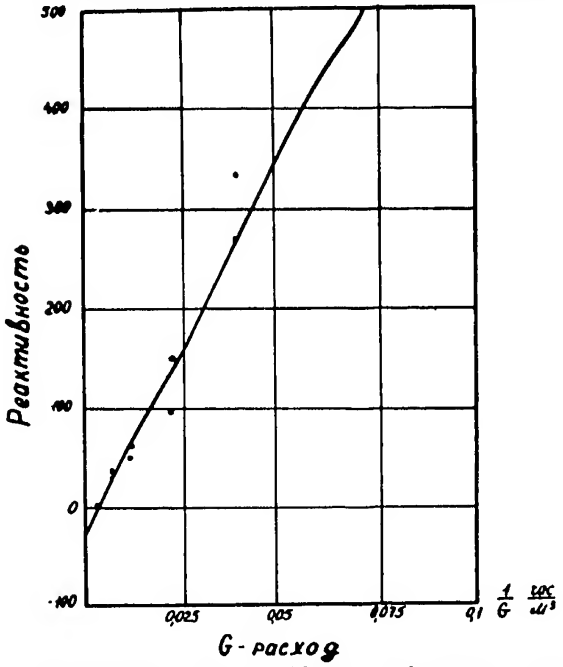


Fig.6 Reactivity change because of changed capacity at various coolant flow rates.
1 - 245m^3/hr; 2 - 145m^3/hr; 3 - 83 m^3/hr;
4 - 44m^3/hr;, 5 - 25m^3/hr; 6 - 10m^3/hr.
Reactivity %. Time - sec.



The dependence of the extreme power effect/cm AR /5mw /from the reversal flow rate.

Рис.

Fig.7 Dependence of the extreme power effect on the value of I/G.
1.Reactivity. 2. G - flow rate.

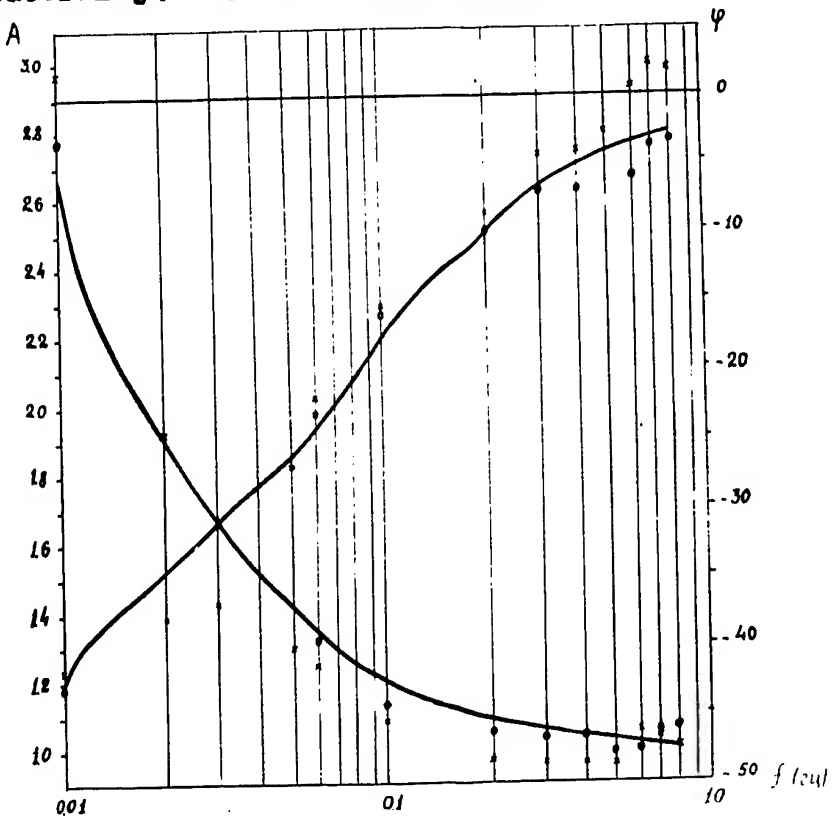


Fig.8 Amplitude and phase characteristics of the reactor at the flow rate of G = 245 m^3/hr.

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